4.0 CONSTITUENTS IN RECYCLED URANIUM

4.1 Analytical Laboratories

4.1.1 Analytical Procedures

Procedures specific to the analytical laboratories were developed to aid personnel in correctly performing various operations. These procedures were primarily to perform various physical operations in the laboratory and included such things as waste management, changing gloves on glove boxes, operation of the ventilation system, etc. The procedures were maintained in a controlled manual.

4.1.2 Analytical Methods

Analytical methods were specific to the particular processes being used and were developed based on standard methods, methods described in the complex literature, and methods described in the open literature. In some cases, the methods were uniquely developed for the special measurements required by the particular process. Each method was placed in a quality control program, then used only by qualified analysts trained in the details of the method. The methods were maintained in a controlled document. Most of the unique methods were used for process control purposes.

4.1.3 Processing Issues

During the first few years of processing, analytical samples were handled with a minimum of shielding and with the manual analytical techniques that were in use at that time. Doses were high while processing samples in that manner. The start up of the Remote Analytical Facility (RAF) relieved some of these issues, but because of the difficulties handling the samples and maintaining the equipment in the facility, many of these issues still remained until the Remote Analytical Laboratory (RAL) was placed into service in 1986.

4.1.4 Quality Assurance

The product solution from the extraction cycles was concentrated to approximately 350 grams per liter and stored in organ pipe banks located in CPP-602. This solution was circulated through the tube banks in an attempt to homogenize the solution. Following denitration in the fluidized bed, each UO_3 product batch was mixed in a V-blender. Samples were taken from the product as it was bottled or placed in the product can. Two samples were sent to the lab for analysis. After the aliquots were taken from the two samples, the samples were blended together, sealed and stored for an archive sample representative of that product batch. Every can or bottle was analyzed for uranium isotopic composition and for total uranium content using isotope dilution mass spectrometry (IDMS). The

U-233 spikes used as the calibration spike in each sample were traceable to the National Bureau of Standards (NBS) and then later National Institute of Science and Technology (NIST) through calibration materials made available by the New Brunswick Laboratory who distributes the radioactive NBS calibration samples.

Every fifth can was analyzed for inorganic and radioactive impurities. The radionuclides included transuranic isotopes, beta emitters, and gamma emitters. The transuranics were typically analyzed using an alpha pulse-height analysis, and the beta emitters were analyzed using a gross beta count. Gamma emitters were analyzed using gamma ray spectroscopy. The labs never specifically analyzed for technetium-99 contamination in the product.

The quality control program at ICPP was based on the routine analysis of matrix matched, blind, control samples. From this data, an estimate of the uncertainty in a measurement could be made. The assumption was that each analyst in the lab would perform like every other analyst. As a result, a single uncertainty estimate was provided with each analytical result based on the statistical data of the whole population in the laboratory. Control samples early in the program were required to be analyzed once per month. After computers came into use, control samples were analyzed on a daily basis for each method used by each analyst. This requirement was enforced through the computer, which would not accept any data from an analyst who did not meet both the precision and bias criteria for that particular analyte. This type of program was an effective daily requalification of the analyst on the methods. The programs in the computer could maintain and update the statistical data, use the statistical data to test the result to determine whether the result was within pre-established specifications, and provides a precision estimate in the form of a single standard deviation value attached to each analytical result for which the statistical data existed.

The control samples and the calibration standards were based on analytical standards available from the New Brunswick Laboratory, who distributed the radioactive standards for the NBS and later the NIST and from NBS for the non-radioactive standards. In some cases, standards were qualified by a round robin of other DOE laboratories. This was particularly true of the isotope dilution mass spectrometry (IDMS) standards used for the accountability measurements of uranium mass and the uranium isotopic distribution.

Sampling was prescribed by specific sampling procedures to ensure that representative samples were obtained. Various techniques were used to determine that a set of samples were from the same well-mixed, homogeneous population that accurately represented the contents of a tank, product bottle, or can of product.

Characterization of the product samples was based on the receiving site's receipt criteria for the product that was in effect at the time. The primary criteria of interest seemed to be the alpha and gamma specifications. The alpha specification limited the amount of higher actinides present in the product while the gamma specification was a measure of the amount of radiation exposure expected by the workers who had to handle the product. Typically, the beta specification was of less interest because the product was handled in equipment or containers that provided shielding for the beta activity.

In addition to the radioactive component specifications there were also specifications on the amount of inorganic impurities that could be present in the product. Until the top water scrub in the third extraction cycle was installed, the ICPP product was always pushing the limit for aluminum. After the top scrub was installed, there were no problems meeting those specifications.

4.2 Neptunium, Plutonium, and Technetium in ICPP Uranium Product as Estimated by ORIGEN2 Calculations.

Because there is little analytical data on final product as a result of the records retention policy, the project resorted to estimating the quantity of plutonium, neptunium, and technetium-99 from radionuclide inventories based on ORIGEN2 code calculations. These calculations provided data on the radionuclide inventory in the dissolver product. Because the interest is on the contaminants in the final product after the fission products have been removed by the solvent extraction train, experimentally-determined decontamination factors were used to convert the calculated dissolver product radionuclide inventory into a final product inventory.

The ORIGEN2 code (Croff, A.G., 1980) is a computer program that is widely used to estimate the fission product inventory of the fuel in a reactor at any time during its lifetime. It is reactor specific and takes into account the neutron spectrum and the cross sections of the various nuclides. It also includes a half-life table to take into account the decay and ingrowth of the various radionuclides. The ORIGEN2 code also provides an estimation of the actinides produced through activation of a fraction of the uranium present.

To estimate the fission product inventory of fuel that is to be processed, a number of assumptions must be made. The first assumptions were for the specific reactors that the fuels were irradiated in. The reactors chosen were reactors that mimicked the fuels that were predominantly processed at ICPP. For the aluminum fuels, an MTR reactor fuel that achieved maximum burnup was chosen. The initial enrichment was 93.15% U-235, and the final enrichment was assumed to be 78.21% U-235. The fission product inventory was aged for 2.8 years, and the calculation assumed one cycle in the reactor.

The second fuel chosen was a generic PWR-type zirconium-clad fuel element with an initial enrichment of 97% U-235 and final enrichment of 78.48%. The neutron spectrum and the cross sections were typical of a fuel irradiated in the PWR reactor. The radionuclide inventory was assumed to have aged for 3.0 years which was assumed to be the age of the fuel at the time of processing.

The final fuel chosen was a stainless steel fuel that was irradiated in the EBR-II reactor. The EBR-II, MARK IA fuel was assumed to have been burned up in a fast reactor flux with the appropriate cross sections. The initial enrichment was assumed to be 52.9% enriched, and the final enrichment was 51.9%. The fission product inventory was aged 3.0 years, which was assumed to be the age of the fuel at the time of processing.

The code was modified to provide the final output in grams of radionuclide per 100 grams of total uranium, (see Table VI) or as curies of radionuclide per gram of total uranium, as shown in Table VII.

TABLE VI

ORIGEN2 Results in Terms of Grams/100grams of Uranium

Mass of Individual Radionuclides in Dissolver Product Normalized to g / 100 g Total Uranium.

Nuclide	Half-Life		Al	Zr	SS
U-232	7.200E+01	yr	3.1E-07	1.3E-06	2.9E-08
U-233	1.592E+05	yr	8.7E-06	2.7E-06	2.4E-06
U-234	2.445E+05	yr	1.3E+00	1.0E-02	5.3E-01
U-235	7.038E+08	yr	7.8E+01	7.8E+01	5.2E+01
U-236	2.342E+07	yr	1.3E+01	2.0E+01	3.4E-01
U-238	4.470E+09	yr	7.9E+00	1.7E+00	4.7E+01
Np-237	2.140E+06	yr	7.8E-01	1.3E+00	2.3E-03
Np-239	2.355E+00	d	5.6E-10	1.9E-11	2.9E-19
Pu-238	8.775E+01	yr	8.1E-02	2.1E-01	1.3E-05
Pu-239	2.413E+04	yr	3.2E-01	3.1E-02	1.4E-01
Pu-240	6.569E+03	yr	5.3E-02	6.4E-03	2.9E-04
Pu-241	1.440E+01	yr	4.4E-02	1.4E-03	3.8E-07
Pu-242	3.758E+05	yr	6.9E-03	2.3E-04	4.4E-10
Am-241	4.322E+02	yr	6.5E-03	2.8E-04	6.4E-08
Am-242m	1.520E+02	yr	2.2E-06	1.9E-06	2.9E-12
Am-243	7.380E+03	yr	6.5E-04	2.2E-05	3.4E-13
Se-79	6.500E+04	yr	1.0E-02	1.7E-02	2.2E-04
Sr-90	2.912E+01	yr	1.2E+00	1.8E+00	1.8E-02
Y-90	6.410E+01	h	3.0E-04	4.6E-04	4.5E-06
Zr-93	1.530E+06	yr	1.5E+00	2.4E+00	2.4E-02
Tc-98	4.200E+06	yr	4.5E-06	8.8E-06	3.9E-08
Tc-99	2.130E+05	yr	1.4E+00	2.2E+00	2.3E-02
Pd-107	6.500E+06	yr	5.5E-02	8.2E-02	1.9E-03
I-129	1.570E+07	yr	2.3E-01	3.6E-01	5.9E-03
Cs-134	2.062E+00	yr	5.6E-02	7.9E-02	2.5E-05
Cs-135	2.300E+06	yr	2.2E-01	1.8E+00	3.4E-02
Cs-137	3.000E+01	yr	2.0E+00	3.0E+00	3.1E-02
Ba-137m	2.552E+00	m	3.0E-07	4.6E-07	4.7E-09
Ce-142	1.050E+11	yr	2.1E+00	3.4E+00	3.2E-02
Nd-144	2.100E+15	yr	2.1E+00	3.8E+00	2.9E-02
Pm-147	2.623E+00	yr	2.6E-01	1.9E-01	5.6E-03
Sm-147	1.070E+11	yr	3.1E-01	5.9E-01	8.2E-03
Sm-148	8.000E+15	yr	1.1E-01	4.5E-01	1.4E-04
Sm-149	1.000E+15	yr	2.5E-02	7.2E-03	6.6E-03

Table VII

ORIGEN Result in Terms of Ci/gU

Activity of Individual Radionuclides in Dissolver Product Normalized to Ci / g Total Uranium.

Nuclide	Half-Life		Al	Zr	SS
U-232	7.200E+01	yr	6.7E-08	2.7E-07	6.2E-09
U-233	1.592E+05	yr	8.4E-10	2.6E-10	2.4E-10
U-234	2.445E+05	yr	8.1E-05	6.6E-07	3.3E-05
U-235	7.038E+08	yr	1.7E-06	1.7E-06	1.1E-06
U-236	2.342E+07	yr	8.1E-06	1.3E-05	2.2E-07
U-238	4.470E+09	yr	2.7E-08	5.6E-09	1.6E-07
Np-237	2.140E+06	yr	5.5E-06	9.1E-06	1.7E-08
Np-239	2.355E+00	d	1.3E-06	4.3E-08	6.8E-16
Pu-238	8.775E+01	yr	1.4E-02	3.6E-02	2.2E-06
Pu-239	2.413E+04	yr	2.0E-04	1.9E-05	8.9E-05
Pu-240	6.569E+03	yr	1.2E-04	1.5E-05	6.6E-07
Pu-241	1.440E+01	yr	4.5E-02	1.5E-03	3.9E-07
Pu-242	3.758E+05	yr	2.6E-07	8.8E-09	1.7E-14
Am-241	4.322E+02	yr	2.2E-04	9.6E-06	2.2E-09
Am-242m	1.520E+02	yr	2.1E-07	1.8E-07	2.9E-13
Am-243	7.380E+03	yr	1.3E-06	4.3E-08	6.8E-16
Se-79	6.500E+04	yr	7.3E-06	1.2E-05	1.5E-07
Sr-90	2.912E+01	yr	1.6E+00	2.5E+00	2.5E-02
Y-90	6.410E+01	h	1.6E+00	2.5E+00	2.5E-02
Zr-93	1.530E+06	yr	3.7E-05	6.1E-05	5.9E-07
Tc-98	4.200E+06	yr	3.9E-11	7.6E-11	3.4E-13
Tc-99	2.130E+05	yr	2.4E-04	3.8E-04	3.9E-06
Pd-107	6.500E+06	yr	2.8E-07	4.2E-07	1.0E-08
I-129	1.570E+07	yr	4.0E-07	6.3E-07	1.0E-08
Cs-134	2.062E+00	yr	7.2E-01	1.0E+00	3.2E-04
Cs-135	2.300E+06	yr	2.5E-06	2.0E-05	4.0E-07
Cs-137	3.000E+01	yr	1.7E+00	2.6E+00	2.7E-02
Ba-137m	2.552E+00	m	1.6E+00	2.5E+00	2.5E-02
Ce-142	1.050E+11	yr	5.0E-10	8.1E-10	7.6E-12
Nd-144	2.100E+15	yr	2.5E-14	4.5E-14	3.4E-16
Pm-147	2.623E+00	yr	2.4E+00	1.8E+00	5.2E-02
Sm-147	1.070E+11	yr	7.0E-11	1.3E-10	1.9E-12
Sm-148	8.000E+15	yr	3.3E-16	1.4E-15	4.3E-19
Sm-149	1.000E+15	yr	6.0E-17	1.7E-17	1.6E-17

The second part of developing the means to estimate fission product and actinide content in the final product at ICPP was to convert ORIGEN2 code calculated values for those radionuclides that would be present in the dissolver product into concentrations that are representative of the final product. To do this, experimentally-determined values for the efficiency of the decontamination of the dissolver product as it passes through the three extraction cycles were used to calculate the expected concentrations of the contaminants of interest.

ORIGEN2 code calculations were completed for fission products and transuranics that would be present in dissolver product from the three fuel processes (aluminum, zirconium, and electrolytic) used at ICPP. By using this classification, the differences that arise because of the processing chemistry and that would affect the decontamination factor could be taken into account. This approach also recognized differences in enrichment and burnup between aluminum and stainless steel. A fourth process at ICPP processed the low-burnup ROVER fuel, which was contact handled before it was charged to the primary burner. Because the aqueous process for this fuel was essentially identical to the zirconium process, it is conservatively assumed to be bounded by the zirconium process. The dissolver product actinide and fission product estimates from the ORIGEN2 calculations were compared with analytical data on dissolver product samples.

The plutonium, neptunium, and technetium data were converted from calculated dissolver product data to final product information by applying decontamination factors (DFs). The DFs were developed for each process and defined as the ratio of the actinide or fission product in the dissolver product to the actinide or fission product in the final product. The decontamination factors could then be used to estimate the final product contaminant concentration values by dividing the dissolver product concentrations of plutonium, uranium, and technetium by the respective decontamination factor.

Final product values for plutonium, neptunium, and technetium were not recorded explicitly during ICPP operations from 1953 through 1992. For Pu, the receiver (generally Y-12) had provided guidance on minimal acceptance limits for product uranium/plutonium alpha ratios. Estimates on the uranium/plutonium product mass ratios can be calculated when the alpha ratio is available. Neptunium limits were not provided by product receivers, and neptunium data is very limited. Technetium was never determined for ICPP uranium product and must be estimated from process decontamination factors for total beta.

The measured alpha ratios (total uranium product alpha/plutonium alpha) for ICPP uranium product was routinely reported (Henry, 1971; Henry, 1973; Wheeler, 1966; Bjorklund, 1974; Bendixsen, 1972; Offutt, 1968; Bendixsen,

1969), and the range of values for a variety of spent fuel types processed could be assessed from a number of published campaign reports. The observed ranges for aluminum, zirconium, and stainless steel are 600-5000, 2000-400,000, and 1000-160,000, respectively. The resulting uranium/plutonium mass ratios in the ICPP product are shown in Table VIII.

The confidence and validity of the product mass ratios can be checked through using measured and recorded decontamination factors for plutonium. The uranium/plutonium mass ratio in the product can be estimated by multiplying the process feed concentrations (fuel dissolver product) with the overall three-cycle decontamination factor. This comparison of two methods for estimating the uranium/plutonium product mass ratio is summarized in Table VIII. It is observed that the U/Pu mass ratio as estimated by the decon factor is consistently lower than that estimated using the alpha ratios. However, as one observes, the two order magnitude variability in alpha ratio and decontamination factor makes a one order of magnitude variability in the comparison less important.

Since the alpha ratio is a more direct product measurement, its uranium/plutonium mass ratio may be considered the more reliable. Table IX lists the contaminant mass ratios which are considered to be a practical maximum for the ICPP product. These values were developed from the ORIGEN2 code calculated values.

Very few neptunium analyses were made in the three-cycle extraction process streams, and no analyses were made for neptunium ICPP uranium product. Some limited data on neptunium decontamination factors are available in the run reports referenced above. From these, a nominal and conservative decontamination factor (product/feed) of 3.2×10^4 has been estimated.

Technetium-99 analyses were never analyzed in ICPP product streams. However, overall beta decontamination factors were measured and documented. The campaign reports consistently noted that ruthenium was the dominant beta emitter with the lowest decontamination factor. Thus, the overall beta decontamination factor for technetium values used in Table IX is confidently believed to be conservative.

Table VIII
COMPARISON OF Pu/U MASS RATIOS
FROM MEASURED DECONTAMINATION FACTORS AND ALPHA RATIOS

	Measured Decontamination Factor for Pu Feed/Product	Measured Alpha Ratio Total Alpha/ U Alpha		Calculated Product Pu/U Mass Ratio, gPu/gU Calculated from ORIGEN2 Code Data, Decontamination Factors	Calculated Product Pu/U Mass Ratio, gPu/gU Calculated from the Measured Alpha Ratio
	Aluminum Clad Fuels			**Aluminum Clad Fuels**	
High	5.0E+03	2.4E+05	Low	1.0E-06	3.0E-09
Median	1.5E+03	5.0E+03	Median	3-4E-06	1.4E-07
Low	6.0E+02	1.0E+03	High	8.4E-08	7.2E-07
	PWR Zirconium Fuels			**PWR Zirconium Fuels**	
High	4.0E+05	5.2E+04	Low	1.0E-06	2.0E-09
Median	8.0E+03	7.3E+03	Median	3.4E-06	1.4E-08
Low	2.0E+03	4.0E+02	High	8.4E-06	2.6E-07
	Stainless Steel Fuels			**Stainless Steel Fuels**	
High	1.6E+05	1.0E+05	Low	8.8E-09	5.2E-07
Median	4.0E+04	1.0E+04	Median	3.5E-08	5.2E-06
Low	1.0E+03	1.0E+03	High	1.4E-06	5.2E-06

Table IX
Contaminants in ICPP Product. Based on ORIGEN2 Code Calculations and DFs from ICPP Process Data

Isotope	Dissolver Product Concentration g/gU	Total Element in Dissolver Product g/gU	Average DF Product/Feed	Product Contaminant Concentration g/gU
		Aluminum Process		
Pu-238	8.1×10^{-4} g/gU			
Pu-239	3.2×10^{-3}			
Pu-240	5.3×10^{-4}	5.0×10^{-3}	6.7×10^{-4}	3×10^{-6}
Pu-241	4.4×10^{-4}			
Pu-242	6.9×10^{-5}			
Np-237	7.8×10^{-3}	7.8×10^{-3}	3.4×10^{-4}	2.5×10^{-6}
Tc-99	1.4×10^{-2}	1.4×10^{-2}	8×10^{-8}	1×10^{-9}
		Stainless Steel Process		
Pu-238	1.3×10^{-7} g/gU			
Pu-239	1.4×10^{-3}			
Pu-240	2.9×10^{-6}	1.4×10^{-3}	2.5×10^{-5}	3.5×10^{-8}
Pu-241	3.8×10^{-9}			
Pu-242	6.9×10^{-12}			
Np-237	2.3×10^{-5}	2.3×10^{-5}	3.2×10^{-4}	7.4×10^{-9}
Tc-99	2.3×10^{-4}	2.3×10^{-4}	8×10^{-8}	2×10^{-11}
		Zirconium Process		
Pu-238	2.1×10^{-3} g/gU			
Pu-239	3.1×10^{-4}			
Pu-240	6.9×10^{-5}	2.5×10^{-3}	1.2×10^{-4}	3×10^{-7}
Pu-241	1.4×10^{-5}			
Pu-424	2.3×10^{-6}			
Np-237	1.3×10^{-2}	1.3×10^{-2}	3.2×10^{-4}	4×10^{-6}
Tc-99	2.2×10^{-2}	2.2×10^{-2}	8×10^{-8}	1.7×10^{-9}

Table IX shows the ORIGEN2 calculated dissolver product data for plutonium, neptunium, and technetium for each of the three main processes. It also shows the decontamination factors and finally the contaminant values for the final product. The total amount of the isotopes of interest can be obtained by multiplying the number of grams shipped by the number of grams of isotope per gram U.

4.3 Analytical Results for Plutonium

4.3.1 Plutonium Specification

The plutonium specification for material to be shipped from ICPP was that the total alpha was not to exceed 5000 dpm/gU. Experimentally, as reported in the Egli report (Egli 1985), the alpha ratio for total transuranics did not exceed 61% and ranged from 31% to 61% of Y-12 informal specification. Since 1977, the alpha ratio has been 31% of Y-12 specification.

4.3.2 Impurity Concentrations for Plutonium in Materials Shipped

Using the data in Table IX, the total plutonium contamination in the final product is 3×10^{-6} g Pu/gU for aluminum fuels, 3.5×10^{-8} gPu/gU for stainless steel fuels, and 3×10^{-7} Pu/gU for zirconium fuels. The decontamination factors used to determine these concentrations are median values from run reports. Some of the plutonium isotope amounts relative to total uranium in the final product are 5.4×10^{-7} g/gU in aluminum product, 3.3×10^{-12} g/gU in stainless steel product, and 2.5×10^{-7} g/gU in zirconium product. For Pu-239 the concentrations in final product are 2.1×10^{-6} g /gU in aluminum product, 3.5×10^{-8} g/gU in stainless steel product, and 3.7×10^{-8} g/gU in zirconium product.

Using the specification of 5000 dpm/gramU a "most probable" result for the alpha contamination can be calculated. These results depend on the isotopic distribution for plutonium from the ORIGEN2 calculation to obtain the most probable value for total plutonium. This calculation produced the result for plutonium which is shown in Table IX. These results are distributed to recognize that the alpha specification is composed of contributions from plutonium and neptunium as well as other higher actinides. The plutonium and neptunium were distributed as a fraction of their mass. Since the alpha specification was at a maximum of 61% of the alpha specification between 1953 and 1976. From 1977 on, the product shipments were 31% of the alpha specification. Thus, there are two entries in the table that distribute the two alpha emitting elements as pre-1976 and post 1976. Because ROVER was a low-burnup fuel, the assumption was made that no significant quantity of plutonium, neptunium and technetium-99 built up in product from this fuel.

Table X shows the total quantities of plutonium, neptunium and technetium-99.

Table XII shows the total quantities of plutonium, neptunium and technetium-99 shipped to the receiving sites.

4.4 Analytical Results for Neptunium in Uranium Materials Shipped

4.4.1 Neptunium Specifications Uranium Materials Shipped

There was no specific neptunium specification other than the general transuranic alpha specification noted above.

4.4.2 Impunity Concentration for Neptunium in Recycled Uranium Shipped

The neptunium plus the plutonium could not exceed 5000 dpm/gU. Since the data in the Egli report indicated that the sum of the neptunium plus the plutonium was consistently below the alpha specification through 1985 and since no modifications were made to the ICPP facility that would adversely affect the decontamination of the alpha emitting transuranic radionuclides, it is expected that this specification which was met for the sum of the amount of plutonium and neptunium, would also be met for neptunium by itself. The neptunium results are also shown in Tables XII, XIII and XIV.

4.5 Analytical Results for Technetium in Uranium Materials Shipped

4.5.1 Technetium Specification in Recycled Uranium

There was no technetium-99 specification in existence during the period that ICPP operated.

4.5.2 Impurity Concentration for Technetium in Uranium Materials Shipped

Since there was no technetium-99 impurity specification for the recycled uranium that ICPP recovered and shipped, there was no attempt made to measure it in the final product. However, it is known that the beta emitter that caused the greatest problem in recycled uranium was ruthenium. It is not expected that the technetium was a significant contaminant in the ICPP uranium product. The technetium results shown in Table XII, XIII, and XIV were calculated from the ORIGEN2 data and the Dfs for technetium-99.

4.6 Analytical Results for Material Received

The ICPP material received was spent fuel. As such, it is out of the scope of this project.

4.7 Discussion of Other Constituents

Because ICPP processed highly-enriched spent fuel, there was a significant amount of isotopes of uranium other than U-238 and U-235 that were produced by the reactor. The U-236 concentration in the final product averaged, 7.6% but

peaked as high as 19.1%. The U-234 concentration averaged approximately 1% but peaked as high as 1.5%.

The uranium-236 content of the fuels varied due to the type of fuel processed. The fuel's uranium-236 content was a function of the burnup and the reactor's neutron spectrum. To determine the average uranium-236 content of the various fuels, analytical data based on the isotopic analyses of monthly composite samples of dissolver product were used. These samples were taken during the operating periods from October, 1980 through November of 1982. The measured uranium-236 were averaged for the specific fuel type and are presented in Table X.

Table X
Uranium-236 Content of ICPP Fuels

Fuel Type	Fuel Quantity Kgs	Average U-236% Content	Range Percent	Total U-236 Kgs
Aluminum	16,147	8.42	6.43 - 11.69	1360
Zirconium	5,468	15.81	13.15 - 19.08	864
Stainless Steel	5,885		1.08 - 1.65	77
ROVER	2,782	0.0	-	0
	30,282 KgsU			2301 KgsU-236

The amount shipped to the various receiving sites and the fuel types they received is shown in Table XI.

Table XI
Uranium-236 Quantities Sent to Receiving Sites

Receiving Site	Uranium Shipped Kgs	Fuel Types Sent	Total U-236 Kgs
Y-12	25,773	Aluminum, stainless steel, zirconium, ROVER	2,227
Portsmouth	4,076	Stainless steel	53
Rocky Flats	219	Aluminum	18
Los Alamos	168	ROVER	0
PNNL	47	Aluminum	4
Totals	30,283		2,302

The range of values is also presented. ROVER fuel was a low burnup fuel and was assumed to have no uranium-236.

Table XII
Concentration of Contaminants in ICPP Product

		Al	Zr	Stainless Steel
1953 - 1976	))Pu	0.043 x 10^{-9} gPu/gU	0.015 x 10^{-9} gPu/gU	21.25 x 10^{-9} gPu/gU
1977 -	)	0.022 x 10^{-9}	0.001 x 10^{-9}	10.80 x 10^{-9}
1953 - 1976	))Np	1187 x 10^{-9} gNp/gU	1633 x 10^{-9} gNp/gu	31.15 x 10^{-9} gNp/gU
1977 -	)	603.3 x 10^{-9}	829.9 x 10^{-9}	15.88 x 10^{-9}
1953 -	)Tc-99	1.1 x 10^{-9} gTc-99/gU	1.8 x 10^{-9} gTc-99/gU	1.8 x 10^{-11} gTc-99/gU

Table XIII
Contaminants in ICPP Product

Al Fuel	Total U Kgs	Plutonium(grams)	Neptunium(grams)	Technetium-99(grams)
1953 - 1976	13,333	5.7×10^{-4}	15.83	0.015
1977 -	2,814	6.2×10^{-5}	1.70	0.003
Zr Fuel				
1953 - 1976	3,082	4.6×10^{-5}	5.03	0.006
1977 -	2,385	2.4×10^{-6}	1.98	0.004
Stainless Fuel				
1953 - 1976	4,508	0.096	0.140	0.0001
1977 -	1,377	0.015	0.022	0.00002
ROVER Fuel	2,783	-	-	-
Total Shipped	30,283 Kgs	0.112 grams Pu	24.70 grams Np	0.028 grams Tc-99
Inventory	1,770	0.019	1.47	0.003
Total Processed	32,053 KgsU	0.131 grams Pu	26.17 grams Np	0.031 grams Tc-99

Table XIV
Material Shipped from ICPP

	Uranium Kgs	Plutonium grams	Neptunium grams	Technetium-99 grams
Portsmouth	4,076	0.087	0.127	0.0001
Y-12	25,773	0.025	24.3	0.028
Rocky Flats	219	0.00001	0.26	0.0002
PNNL	47	0.00000	0.03	0.0001
LASL	168	-	-	-